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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

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10 CFR 50.59
10 CFR 72.48

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
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PALISADES NUCLEAR PLANT
DOCKET 50-255
LICENSE No. DPR-20
REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS AND SUMMARY OF COMMITMENT CHANGES

Nuclear Management Company, LLC, is providing the Palisades Nuclear Plant Report of Facility Changes, Tests and Experiments for the calendar year 2002 and calendar year 2003 (through September 30, 2003). The report is being submitted in accordance with the requirements of 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2) and under the previous requirements 10 CFR 50.59(b)(2). Also included is a summary of revised regulatory commitments as required by Nuclear Energy Institute (NEI) Guideline NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 1, endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff."

Attachment 1 contains a brief description of changes to the facility and a summary of the evaluation, performed in accordance with 10 CFR 50.59 or 10 CFR 72.48, for each. Because of the changes in 10 CFR 50.59, this report is reduced in size from previous reports. There was one change analyzed under the previous requirements of 10 CFR 50.59(b)(2). There were no changes made under 10 CFR 72.48 during this period.

Attachment 2 contains summaries of regulatory commitment changes requiring NRC notification, including justification for the change.

This letter contains no new commitments.

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Attachments

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ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES NUCLEAR PLANT
DOCKET 50-255

OCTOBER 23, 2003

REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS

8 Pages Follow

SDR Review Number: 02-0786 **Document Number:** EA-RCH-02-03

Title: Palisades Small Break LOCA Analysis Acceptance

The current small break loss-of-coolant accident (SBLOCA) analysis of record, described in Final Safety Analysis Report (FSAR) section 14.17.2.2.1 used CEFLASH-4A (A Combustion Engineering methodology). A new analysis was performed by Framatome ANP (Advanced Nuclear Power) using the Nuclear Regulatory Commission (NRC) approved S-RELAP analysis methodology (EMF-2328(P)(A), PWR Small Break LOCA Evaluation Model) to perform the SBLOCA analysis for Palisades. The analysis demonstrated that the 10CFR50.46(b) criteria are satisfied for a SBLOCA and supports operation at a reactor power level of 2580.6 MWt. The Framatome SBLOCA analysis does not adversely affect accidents or malfunctions nor does it create a new type of event not previously evaluated in the FSAR. Because the calculated peak cladding temperature (1,898°F) is below the 10CFR50 Appendix K limit of 2,200°F, the design limit for the fission product barrier was not exceeded.

However, this evaluation concluded that the new analysis is a "departure from a method of evaluation" under 10CFR50.59, and that NRC approval is required prior to implementation. A submittal to the NRC to add the NRC approved S-RELAP analysis methodology to the Core Operating Limits (COLR) section of the Technical Specifications was submitted in correspondence dated January 28, 2002. This was approved by the NRC in a safety evaluation (SE) for Amendment 209 dated 9/13/02.

SDR Review Number: 02-1176 **Document Number:** EA-LOCA-2001-01

Title: Containment Response to a LOCA Using CONTEMPT-LT/28 (Revised SDR #01-1441)

This is a revision to the original 10CFR50.59 Screen and Evaluation (01-1441) that was initiated as a result of a 10CFR50.59 self assessment. The changes consisted of emphasizing that the changes to the analyses involve analytical input parameter changes rather than methodology changes, clarifying the description of emergency operating procedure (EOP) guidance concerning the isolation of containment spray flow in an accident scenario, and several minor enhancements and clarifications. None of the changes affect the conclusions of the 10CFR50.59 screen and evaluation, or the bases for the conclusions. The original plant review committee (PRC) summary, which did not require revision, follows:

Engineering Analyses EA-LOCA-2001-01 and EA-MSLB-2001-01 address the impact of containment spray valve modification EAR-2000-0302 on containment pressure and temperature for a LOCA and a main steam line break accident (MSLB). The modification adds an automatic function to align high pressure safety injection suction sub-cooling upon receipt of a recirculation actuation signal (RAS), provides the ability to manually override a containment high pressure signal to close a containment spray valve to maintain adequate net positive suction head if only one containment spray pump is operating, and provides circuitry to alter operation of selected valves if a containment sump outlet valve (CV-3030) fails to open.

Because it may be advantageous to isolate a spray header prior to RAS, plant procedures are being revised to provide for this. This would result in a slight increase in the containment building temperature and pressure during long term heat removal from containment. Under 10CFR50.59, this is considered to be an adverse change and a 10CFR50.59 evaluation was required.

The evaluation concluded that NRC approval is not required for this change. In both the LOCA and MSLB accidents, the increase in containment temperature and pressure due to decreased spray flow occurs after the peak temperature and pressure, so containment design temperatures and pressures are not exceeded. Moreover, the increased containment temperature is still bounded by the containment environmental qualification temperature profile (EA-BHS-EQ-2001-02). Based on this, the impact of the changes on the accident analyses is less than minimal and the consequences of an accident or malfunction are not increased.

This 10CFR50.59 Evaluation addresses only the effect of the modification on containment temperature and pressure. Other effects are addressed under EAR-2000-0302 (Log numbers SDR-01-1422 and SDR-02-1148).

SDR Review Number: 02-1148 **Document Number:** EAR-2000-0302

Title: Revised - Installation of CHP Bypass for Containment Spray Valves CV-3001 & CV-3002 and RAS Actuation of HPSI Subcooling Valves CV-3070 & CV-3071

This is a revision to 10CFR50.59 Evaluation 01-1422. It was the result of a self assessment where the assessor felt the 10CFR50.59 Evaluation was weak in describing procedures and training. The original write up was enhanced with more detailed description of procedure changes and training. The original PRC summary was not changed by the extra detail. The original is included below:

This modification addresses instances in which the high pressure safety injection (HPSI) and containment spray pumps may have inadequate net positive suction head (NPSH) during a LOCA scenario. The changes involve 1) adding the ability, through the installation of key lock switches, to close one of the containment spray valves, 2) changing circuitry to automatically open the sub-cooling valves to operating HPSI pumps upon receipt of a RAS, and 3) adding circuitry that senses the failure of a sump outlet control valve to open and then closes a containment spray valve and blocks opening of a sub-cooling valve.

This modification potentially affects the design functions performed by the engineered safeguards system in an adverse manner. Therefore, the modification requires evaluation under 10CFR50.59.

Two of the changes involved replacing a previous operator manual action with an automatic action. Specifically, an existing procedurally controlled temporary alteration for closing one of the spray valves is being replaced with a key switch. The alignment of containment spray for HPSI sub-cooling is being changed from a manual action to an automatic action. By replacing required operator manual actions with automatic actions, these changes are deemed to be less than minimal under 10CFR50.59 because they reduce the likelihood of a malfunction due to operator error while responding to plant conditions in a LOCA scenario. Therefore NRC approval is not required.

The modification also assures that failure of a containment sump control valve to open upon a RAS will not result in inadequate NPSH to the spray pumps. Circuitry is being added to detect failure of the control valve to open and then isolate a containment spray header and block opening of a sub-cooling valve. These actions reduce the spray pump required NPSH such that sufficient NPSH is available. This change reduces the likelihood of a malfunction of the containment spray pumps due to failure of a sump control valve to open. Therefore, NRC approval is not required.

One of the effects of isolating a containment spray header is that containment spray flow is reduced. This affects the LOCA containment response and dose consequences analyses. These analyses were revised to reflect the reduced containment spray flow rate and were evaluated under 10CFR50.59. In all cases, the effect of reduced spray flow was found to be less than minimal and NRC approval was not required. The relevant analyses were the LOCA, maximum hypothetical accident/control rod ejection, and MSLB analyses. The 10CFR50.59 evaluations were log numbers 01-1441, 01-1448, and 01-1459.

SDR Review Number: 03-0605 **Document Number:** EDC-TM-2002-022-01

Title: Engineering Design Change to Address Revision of Expected Removal Date which Impacts Basis of Temporary Modification

The static line over the 345 kV transmission lines connecting the plant main transformer to the switchyard has broken. It was proposed to not restore the portion of the wire between the break and the switchyard, and return to power operations without the benefit of this static line. The portion of the wire between the break and the plant will remain installed. The break is a very short distance, approximately one foot, from the line's connection to a tower. This condition will exist from December 2002 until the refueling outage in March 2003. The transmission towers between the plant and the switchyard hold two sets of transmission lines. One set connects the main generator output to the transmission grid through the main step-up transformer and the switchyard. The other set provides start-up power to the plant from the switchyard rear (R) bus. The static line above the main power transmission lines is broken, while the static line above the start-up power lines is intact.

The evaluation concludes that the increase in frequency of accidents and the increase in likelihood of malfunctions due to the removal of a section of one of the two static wires are less than minimal. There are no consequences, fission product boundaries, or methodologies involved. No new accidents or malfunctions are created. The removal of a section of one static wire reduces the depth of lightning protection and increases the fault impedances (affects protective relaying). Neither of these effects is more than minimal. The wire being removed is only a part of these protective features. And finally, the probability of lightning during the period in question (December to March) is very low (0.3% of the total annual lightning strikes).

This engineering design change amends the above description to point out that the static line for the start up lines is still intact and providing adequate protection (during the outage) for required offsite power. This was added when the outage schedule pushed the static line replacement into April.

SDR Review Number: 03-0702 **Document Number:** EDC-FC-978-01

Title: Replace Damaged Fuel Assembly S-10 with Fuel Assembly R-46

This evaluation is an update to the FC-978 revision 0 evaluation contained in SDR-02-0536 and FC-978 revision 1 evaluation contained in SDR-03-0377. The updated evaluation addresses replacing the damaged S-10 assembly with the fuel assembly R-46. It also updates some references to reflect updated analyses and procedures. The conclusion and the bases for the conclusion contained in the previous evaluations have not changed.

SDR Review Number: 02-0536 **Document Number:** FC-978

Title: Conceptual Design of the Modification Package for Cycle 17 Reload Design

The Cycle 17 reload will consist of 64 fresh batch "U" assemblies, 56 once-burnt batch "T" assemblies, 56 twice-burnt batch "S" assemblies, 20 thrice-burnt batch "R" assemblies, 4 batch "U" shield assemblies and 4 Shield Assembly - N (SAN) assemblies. The cycle 17 core design is similar to the cycle 16 core design. It will not adversely affect the FSAR design function of providing rated thermal power without exceeding fuel damage limits of excessive temperature, cladding strain and cladding stress during normal operating conditions and anticipated transients. As a result, there is no more than a minimal increase in the frequency of occurrence or consequences of an accident or malfunction previously evaluated in the FSAR. The functions and interfaces of the core have not been changed. As a result, there is no possibility of creating an accident of a different type than any previously evaluated in the FSAR. The mechanical design of the core is basically unchanged. Hence, the possibility of a malfunction of an system, structure or component (SSC) important to safety with a different result than any evaluated in the FSAR is not created. Also, the design basis limit for a fission product barrier is not exceeded or altered. Two new analysis methodologies are used for the cycle 17 safety analyses. These are the SBLOCA and non-LOCA methodologies. Both of these methodologies have received NRC approval (SE for Amendment 209 dated 9/13/02) for use at Palisades. As a result of the above discussion, further NRC review and approval of the cycle 17 core design is not required prior to implementation of the project.

SDR Review Number: 02-1174 **Document Number:** FC-974 / 50.59

Title: Reload "T" for Fuel Cycle 16

This is a revision to Safety Evaluation 01-0691. It was the result of corrective action CAP031835, which identified an editorial error in that the evaluation had a wrong reference.

SDR Review Number: 03-0377 **Document Number:** FC-978

Title: FC-978 Screen and Evaluation Revision 1 - Detailed Design of the Modification Package for Cycle 17 Reload Design

This evaluation is an update to the FC-978 revision 0 evaluation contained in SDR-02-0536. This evaluation enhances the description of the Batch U shield assemblies and the burnup dependent peaking factor. It also updates some references to reflect the final core design reports. The conclusion and the bases for the conclusion contained in the previous evaluation have not changed.

SDR Review Number: 03-1002 **Document Number:** FC-978 50.59 REVISION

Title: Detailed Design of the Modification Package for Cycle 17 Reload Design

This evaluation is an update to the FC-978 revision 0 evaluation contained in SDR-02-0536, the FC-978 revision 1 evaluation contained in SDR-03-0377 and the FC-978 revision 2 evaluation contained in SDR-03-0702. The updated evaluation incorporates FSAR changes. The conclusion and the bases for the previous evaluations have not changed.

SDR Review Number: 99-1335 **Document Number:** FSAR-1877

Title: FSAR Chapter 9, Table 9-1

This FSAR change revises FSAR Table 9-1 to reflect the correct value for service water flow to the diesel generators. The flow value was based on diesel operation at 2,500 kW. The revised value reflects service water flow requirements for a lake temperature of 85 deg. F and 2750 kW. (Reference C-PAL-99-1478)

The change does not reflect an unreviewed safety question since the higher flow value is within the design capabilities of the diesel engine cooling system as well as the actual, preset service water flow value to the diesels. This FSAR change had been pending (while some outstanding comments were resolved) and used the old 10CFR50.59 process which is a conservative approach in place of initiating a new screening document.

SDR Review Number: 02-1203 **Document Number:** TM-2002-022

Title: Temporary Modification to Remove the Failed Static Line from the Plant Output Transmission Lines

The static line over the 345 kV transmission lines connecting the plant main transformer to the switchyard has broken. It was proposed to not restore the portion of the wire between the break and the switchyard, and return to power operations without the benefit of this static line. The portion of the wire between the break and the plant will remain installed. The break is a very short distance, approximately one foot, from the line's connection to a tower. This condition will exist from December 2002 until the refueling outage in March 2003. The transmission towers between the plant and the switchyard hold two sets of transmission lines. One set connects the main generator output to the transmission grid through the main step-up transformer and the switchyard. The other set provides start-up power to the plant from the switchyard rear (R) bus. The static line above the main power transmission lines is broken, while the static line above the start-up power lines is intact.

The evaluation concludes that the increase in frequency of accidents and the increase in likelihood of malfunctions due to the removal of a section of one of the two static wires are less than minimal. There are no consequences, fission product boundaries, or methodologies involved. No new accidents or malfunctions are created. The removal of a section of one static wire reduces the depth of lightning protection and increases the fault impedances (affects protective relaying). Neither of these effects is more than minimal. The wire being removed is only a part of these protective features. And finally, the probability of lightning during the period in question (December to March) is very low (0.3% of the total annual lightning strikes).

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES NUCLEAR PLANT
DOCKET 50-255

OCTOBER 23, 2003

SUMMARY OF COMMITMENT CHANGES

2 Pages Follow

ATTACHMENT 2
SUMMARY OF COMMITMENT CHANGES

COMMITMENT NUMBER	SOURCE DOCUMENT/DATE	COMMITMENT DESCRIPTION	REVISED COMMITMENT	JUSTIFICATION
1011446	Bulletin 85-01 response February 26, 1986	Procedural controls should remain in effect until completion of hardware modification to substantially reduce the likelihood of steam binding or until superceded by action implemented as a result of resolution of Generic Issue 93. Feb 26, 1986 letter commits that: A requirement for monitoring the auxiliary feedwater piping has been added to the "Auxiliary Building Plant Data Sheet No. 2." This data sheet is completed by the Auxiliary Operator each shift (8 hours) and is reviewed and signed by the Shift Supervisor.	A requirement for monitoring the auxiliary feedwater piping has been added to the operator rounds electronic database. The data may be taken electronically or recorded on paper copies of the data log sheet. The data is taken by the Auxiliary Operator at approximately 12-hour intervals and is reviewed and approved by a licensed operator.	1) The level of review is not material to the successful completion of the monitoring of AFW piping. 2) The change in the monitoring frequency from every eight hours to every twelve hours represents an additional four hours in which to recognize valve leakage. The four-hour increase in the time that an unrecognized inoperability could be present on one train of AFW is well within the 72 hour LCO action statement time that is allowed by Technical Specification 3.7.5.
1014677	Generic Letter 88-03 response May 9, 1988	Maintain procedure to AFW system fluid conditions and recognize steam binding. May 9, 1988 letter commits to the following: As stated in our response to IE Bulletin 85-01 dated Feb 26, 1986, the requirement to monitor the Auxiliary Feedwater piping is a part of our Auxiliary Building Data Sheet (Primary). This data sheet is completed by the Auxiliary Operator each shift and is reviewed and signed by the shift Supervisor.	A requirement for monitoring the auxiliary feedwater piping has been added to the operator rounds electronic database. The data may be taken electronically or recorded on paper copies of the data log sheet. The data is taken by the Auxiliary Operator at approximately 12-hour intervals and is reviewed and approved by a licensed operator.	1) The level of review is not material to the successful completion of the monitoring of AFW piping. 2) The change in the monitoring frequency from every eight hours to every twelve hours represents an additional four hours in which to recognize valve leakage. The four-hour increase in the time that an unrecognized inoperability could be present on one train of AFW is well within the 72 hour LCO action statement time that is allowed by Technical Specification 3.7.5.
1012465	Confirmatory Action Letter Response December 1, 1986	Material Condition Task Force – CIS-03: Generic Issue – Containment isolation valves: local leak rate test (LLRT) program – implement an augmented LLRT program which will increase frequency for testing all penetration valves.	This commitment is being removed from resident or ongoing status and is being closed.	Technical Specification SR 3.6.1.1 requires that containment leakage rate testing of valves be performed in accordance with the containment leak rate testing program. Technical Specification ADMIN 5.5.14, Containment Leak Rate Testing Program, requires that a testing program be established in accordance with 10 CFR 50, Appendix J, Option B. Compliance with Technical Specification SR 3.6.1.1 and ADMIN 5.5.14 fulfill this commitment.

ATTACHMENT 2
SUMMARY OF COMMITMENT CHANGES

COMMITMENT NUMBER	SOURCE DOCUMENT/DATE	COMMITMENT DESCRIPTION	REVISED COMMITMENT	JUSTIFICATION
1012466	Confirmatory Action Letter Response December 1, 1986	Material Condition Task Force – CIS-03: Generic Issue – Containment isolation valves: local leak rate test (LLRT) program – develop valve trending program to track valve performance for identification of degradation prior to valve failing leak rate.	This commitment is being removed from resident or ongoing status and is being closed.	Technical Specification SR 3.6.1.1 requires that containment leakage rate testing of valves be performed in accordance with the containment leak rate testing program. Technical Specification ADMIN 5.5.14, Containment Leak Rate Testing Program, requires that a testing program be established in accordance with 10 CFR 50, Appendix J, Option B. Compliance with Technical Specification SR 3.6.1.1 and ADMIN 5.5.14 fulfill this commitment.
2011097	NRC Bulletin 80-10 response July 8, 1980	The air receiver tanks will be sampled and gamma analysis performed weekly. This will be implemented by 10/1/80 due to potential complications in ability to obtain a sample.	The commitment remains the same, except the frequency of the sampling is being changed from weekly to quarterly per HP 6.51.	A change in the frequency was determined to be appropriate based on acceptable past sample results.